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A SUMMARY OF MODIFICATIONS TO THE ORIGINAL EBR-II REACTOR SHUTDOWN SYSTEM

by

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A SUMMARY OF MODIFICATIONS TO THE ORIGINAL EBR-II REACTOR SHUTDOWN SYSTEM

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I. ABSTRACT

This report describes and summarizes those modifications made to or proposed for the original[1,2] EBR-II reactor shutdown system (RSS) and reactor building containment isolation system.* It further provides a history of the formal Plant Protection System (PPS) upgrading effort, the rationale behind the effected and proposed changes, and a summary of supporting safety analyses. By so doing, it provides a convenient form by which the overall performance characteristics of the modified RSS may be seen.

*The Plant Protection System (PPS), also called the Safety System (SS), is made up of the Reactor Shutdown System (RSS) and the Engineered Safety Features (ESF). The containment isolation system is an ESF and is included here because it was originally part of the RSS.

II. INTRODUCTION

In the late 1960's, the EBR-II experimental program work was redirected to utilize its potential as a LMFBR fuels irradiation facility. This programmatic shift prompted the initiation of a continuing effort to upgrade the original RSS, thus minimizing reactor downtime due to spurious trips. To this end, detailed safety analysis has been provided in support of a number of deletions and revisions to the RSS. Specifically, of the 69 original trip and permissive interlock functions listed in Table III of the Hazard Summary Report (HSR) Addendum,[2] 24 have been deleted from the RSS, two have been converted to permissive interlocks for reactor startup, and three have been eliminated with installation of the wide-range nuclear channels. One function added to the RSS subsequent to the HSR Addendum has also been deleted. Two of the four original isolation trip parameters are proposed for deletion; two other functions are proposed for addition as isolation trips. In addition, those circuits identified both in the HSR and subsequent safety analysis as being required to protect against identified faults have been modified to upgrade performance and reliability. These changes and others proposed are summarized in Table I. Documents containing supporting safety analyses for these changes are listed in Table II.

TABLE I. Summary of RPS Upgrading Effort

Trip Parameter	Action	When Completed	Plant Modification No.	Supporting Document
Shutdown string for reactor operate mode	Added second redundant shutdown system (System B) for required parameters	7/75	796	ANL-76-34
Earthquake	Replaced detector with 3 detectors (horizontal and vertical) in diverse locations.	11/71	396	----
	Addition to isolation trip circuit	Pending	WAF-5137	ANL-76-33 ANL-76-34
Primary flow low: High pressure plenum No. 1 & 2 Low pressure plenum No. 1 & 2 Total reactor flow	Deleted high pressure plenum and rate-of-change trips for pump No. 1	<7/70	---	ANL-7743
	Upgraded circuitry	5/75	443	ANL-76-31 ANL-75-40 ANL-76-34
	Replaced total reactor flowmeter 514E with flow system 541E	10/76	WAF-5099	ANL-76-34
2400-V undervoltage pump trip	Added to RSS	<7/70	---	ANL-7743
	Upgraded	10/72	410	----
Low current to primary pumps	Added	<7/70	---	ANL-7743

Trip Parameter	Action	When Completed	Plant Modification No.	Supporting Document
Primary pump trips (both pumps): Generator output breaker open Low current High motor winding temperature Low clutch and brake cooling water pressure Low MG set clutch voltage MG set supply voltage breaker	Deleted	5/75	443	ANL-76-34 ANL-76-31 ANL/EBR-018
Subassembly outlet temperature high	Upgraded	5/75	443	ANL-76-34 ANL-76-31
Reactor outlet temperature high	Deleted	2/74	784	----
Source flux level low Period short Count rate high (fuel handling) Power level high Linear power level high Auto flux control rod down High voltage low	Deleted auto flux control rod down	<5/70	178C	ANL-7743
	Replaced Ch. 1,2,3,7,9,10,11	2/69	200	----
	Added Ch. 7A	3/69	230	----
	Upgraded Ch. 1,2,3,4,5,6	12/69	222	----
	Replaced Ch. 4,5,6	7/72	398	----
	Replaced Ch. 1,2,3,4,5,6,9,10,11 with wide range channels A,B,C	6/75	WAF-753	ANL-76-34 ANL-76-32 ANL-75-41
	Deleted automatic power level trip	6/75	WAF-753	ANL-76-34
	Addition of power level high isolation trip circuit	Pending	WAF-5137	ANL-76-33 ANL-76-34

Trip Parameter	Action	When Completed	Plant Modification No.	Supporting Document
Reactor building isolated: (Activated by the following isolation parameters: Radiation level high Reactor building air temperature high Reactor building air pressure high Subassembly outlet temperature high)	Deleted	4/76	WAF-5088	ANL-76-34 ANL/EBR-072
	Upgrading of isolation trips (includes deletion of reactor building air temperature and pressure high trips)	Pending	WAF-5137	ANL-76-34 ANL-76-33 ANL/EBR-040
Instrument thimble temperature high	Modified trip sensors and circuit logic	<5/70	---	ANL-7743
	Deleted	∿4/73	405	----
Any control rod unlatched	Converted to permissive interlock for reactor startup	7/75	404	ANL-76-34
Safety rods not fully up	Installed bypasses for use during Sequences A and H of unrestricted fuel handling and seal cleaning	2/69 11/72	226 338	---- ----
Bulk sodium temperature high Bulk sodium level high	Replaced level sensor	12/69	231	----
	Replaced temperature thermocouples	4/70	264	----
	Deleted	10/75	WAF-5087	ANL-76-34
Argon cover gas temperature high Argon cover gas pressure high	Deleted	7/75	WAF-5069	ANL-76-34

Trip Parameter	Action	Completed	Plant Modification No.	Supporting Document
FERD count level high FERD high voltage low	Added both parameters	<5/70	---	ANL-7743
	Deleted both parameters	1/77	WAF-5144	ANL-76-34 ANL-75-94
Fuel handling incomplete:	Added cable connectors and INCOT	<5/70	---	ANL-7743
Sequences A and H complete Control power off Cable connectors connected INCOT I-III down	Converted to permissive interlock circuit for reactor startup *Retained as scram parameters	6/74	431	ANL/EBR-062
*Reactor vessel cover down *Reactor vessel cover locked *Reactor vessel cover holddown springs down	Removed cable connectors for INCOT's	4/76	WAF-5130	----
Crane position satisfactory (startup permissive)	Added	<5/70	---	

TABLE II

Documents Containing Supporting Safety Analysis for

PPS Upgrading Effort

Document Number	Title	Authors	First Written	Latest Revision
---	FSAA for PM WAF-796, Upgrading of Shutdown String*	---	---	6/75
---	FSAA for PM 443, Upgrade Loss of Flow Protection*	---	---	12/75
---	FSAA for PM WAF-5099, Total Primary Flow Measurement Channel, FT-541E**	---	---	12/76
---	FSAA for PM WAF-753, Upgrade Reactivity Protection*	---	---	11/75
---	FSAA for PM WAF-5088, Remove Reactor Building Isolation Trip from RSS*	---	---	4/76
---	PSAA for PM WAF-5137, Modification of Reactor Building Isolation System	---	---	3/77
---	FSAA for PM WAF-404, Removal of the Control Rod Not Latched Trip Contacts from the Shutdown Circuit*	---	---	1/75
---	FSAA for PM WAF-5087, Removal of Bulk Sodium Temperature and Level Trips from RSS*	---	---	10/75
---	FSAA for PM WAF-5069, Removal of Cover Gas Temperature and Pressure Trips from RSS*	---	---	9/75
---	FSAA for PM WAF-5144, Remove FERD Trip from the RSS for Operation with Breached Fuel Experiments (Including RBCB and CODE Programs)**	---	---	3/77
---	Engineering package for PM WAF-405, Removal of Instrument Thimble Temperature High Trip from the Shutdown Circuit	---	---	1/73

*To be published in ANL-76-34, Vol. I.

**To be published in ANL-76-34, Vol. II.

Document Number	Title	Authors	First Written	Latest Revision
ANL/EBR-018	Surveillance and Evaluation of the EBR-II Flow Monitoring System	R.O. Haroldsen et al.	---	4/70
ANL/EBR-040	Evaluation of the EBR-II Reactor Building Isolation System	R.O. Haroldsen	---	6/71
ANL/EBR-046	Design Basis Document for Trips Related to Loss of Primary Flow in the EBR-II Plant Protection System (PPS)	J. F. Boland et al.	∿ 1972	7/74
ANL/EBR-048	Design Basis Document for Reactivity Change Related Trips in the EBR-II PPS (in the Operate Mode)	R. N. Curran et al.	∿ 1972	7/74
ANL/EBR-062	An Evaluation of the EBR-II Fuel Handling Handling Console Circuit	R.O. Haroldsen R. N. Curran	3/72	2/74
ANL/EBR-072	Basis for the EBR-II Reactor Containment Isolation Criteria	J. A. Bjorkland, W. F. Booty	∿ 1972	12/75
ANL-7665	Study of the Response of the EBR-II Plant Protective System to Hypothetical Malfunctions in the Reactor System	A. V. Campise	---	6/70
ANL-75-40	Response of EBR-II to Off-Normal Primary-Coolant Flow	E. M. Dean J. I. Sackett	12/74	8/76
ANL-75-41	Response of EBR-II to Reactivity Insertion	E. M. Dean J. I. Sackett	8/74	To be published
ANL-76-31	Design Basis Document for Trips Related to Loss of Primary Flow in the EBR-II Plant Protection System (PPS)	J. F. Boland et al.	7/74	To be published
ANL-76-32	Design Basis Document for Reactivity Change Related Trips in the EBR-II PPS (in the Operate Mode)	R. N. Curran et al.	7/74	To be published

Document Number	Title	Authors	First Written	Latest Revision
ANL-76-33	Basis for the EBR-II Reactor Containment Isolation Criteria	C. C. Price et al.	12/75	To be published
ANL-76-34	Final Safety Analysis Addenda for EBR-II Hazard Summary Report: Plant Protection System Upgrading, Vols. I and II	---	---	To be published
ANL-76-94	The Efficacy of the EBR-II FERD System as an Automatic PPS Device	R. M. Fryer et al.	11/76	To be published
ANL-77-32	Safety Considerations for Reactivity Change Related Trips in the EBR-II PPS (in the Fuel Handling Mode)	H. A. Larson J. I. Sackett	5/77	To be published

III. HISTORY OF PPS UPGRADING EFFORT

A. Philosophy Relative to Protective Functions

The original role for EBR-II was that of a power-generating experimental facility with an integrated fuel cycle and a limited lifetime. As such, it appears to have been very conservatively designed with regard to the inclusion of trip parameters in the RSS. This contention is supported by the fact that 33 of the 69 trip parameters listed in Table III of the HSR Addendum[2] were not indicated as required in Table VII of the HSR.[1] In other words, between 1957 and 1962, 33 additional trip parameters were added to the RSS; and several more were added later.[3] Further, the majority of these trip parameters are not required by any of the detailed safety analysis of either the HSR or Addendum, but are merely listed as part of the RSS. Indeed, the safety analysis in support of the EBR-II Technical Specifications requires only 18 of the 69 trip parameters listed in the HSR Addendum. Because previous experience was limited for design of the RSS, the philosophy was adopted that where there was a potential utility of a trip function, it was added. As a result, spurious trips during both reactor operation and fuel handling were frequent.

As long as EBR-II was operated as an experimental facility with a limited lifetime, an excessive number of spurious trips could be tolerated. However, when EBR-II's mission was changed to that of a fuels irradiation facility of maximum lifetime, the negative effect of spurious trips on both plant factor and plant system lifetime could no longer be ignored. A reduction in the total number of spurious trips became a major goal for EBR-II, and a major analytical effort was launched to determine which trip parameters were needed for safety and which were not. At the same time, engineering effort was directed toward upgrading the system circuitry and components insofar as feasible along the guidelines established in RDT Standard C16-1T.[4] This philosophy with regard to trip parameters thus changed from one of extreme conservatism without supporting analysis to one of defining functional requirements with supporting analysis. Those that were not required for safety could be eliminated, and those that were required would be upgraded. This philosophy has led to the various plant modifications described in subsequent sections of this report and to the continuous and ongoing effort to upgrade the EBR-II PPS.

One further note: Prior to the issuance of RDT Standard C16-1T, which defined the PPS in terms of both the RSS and the Engineered Safety Features, the words "reactor shutdown system" and "Plant Protection System" were synonymous and interchangeable. Further, the Engineered Safety Features were not identified for EBR-II until late in 1975, only a few months before the initial PPS upgrading effort was completed.

B. Chronology of Events

In a TWX dated January 8, 1969, directed to M. Levenson, EBR-II Project Director, Milton Shaw, Director of the Division of Reactor Development and Technology (RDT), USAEC, "requested ANL to conduct a critical review, on a priority basis, of the EBR-II scram instrumentation." An exchange of correspondence between RDT and ANL culminated in a joint meeting in June 1969 in which preliminary guidelines for the study were defined. The authority to proceed with the study at a rate of $50K a year came from Shaw in August 1969 and a preliminary study was begun.

In October 1969, J. F. Boland, of ANL-West at that time, was assigned to do an independent study of the system. This study was completed in April 1970 and made the following general recommendations:

1. That the trips required to protect against incidents analyzed in the HSR[1] and Addendum[2] (power level, period, flow, and subassembly outlet temperature) be evaluated and upgraded; and

2. That the criteria of the newly issued RDT Standard, C16-1T, "Supplementary Criteria and Requirements for RDT Reactor Plant Protection Systems," be used as the basis for removal of a number of anticipatory trips.

The study also evaluated each type or category of reactor trip (protective subsystem) and made specific recommendations for improvements and deletions.

A formal plan and schedule were agreed upon in April 1971. At that time, the highest priority was placed on the upgrading or deletion of trips that were the source of the greatest number of spurious trips. Also, each protective subsystem was categorized as either "essential," "aid and comfort," or "nonessential." Table III shows the categorization and priorities established.

Based on the above, in June 1971 a work project request was initiated for funding of the physical modifications required to upgrade the PPS. Work Project 1011 was established in September 1971 to fund the project; the analytical support effort was funded directly from operating funds (ANL Task 46).

In July 1971, the rationales for the deletion of the following eight trip functions were provided to RDT:

Control rod not latched
Instrument thimble temperature monitor
Reactor plant isolation trip
Bulk sodium level
Argon blanket temperature
Argon blanket pressure
FERD trip
Control rod air assist pressure

TABLE III. PROTECTIVE SUBSYSTEM CATEGORIES AND SCOPE OF UPGRADING EFFORT, APRIL 1971

Protective Subsystem	Priority	Category*	Analysis Required	Proposed Action
Earthquake detection	1	A	Seismic study	Upgrade
Channels 4, 5, 6 period	1	B	DBD	TBD
Channels 1, 2, 3 period, level (FH)	1	A	DBD	Upgrade
Coolant flow rate of change	1	C	SA	Delete
FERD Channels A, B, C	1	C	SA	Delete
Primary pumps 1 and 2	1	C	SA	Delete
Reactor building temperature	1	B	DBD	TBD
Reactor building isolation	2	C	SA	Delete
Control rod manual trip	2	A	SA	Upgrade
Argon blanket pressure	2	B	SA	Delete
Argon blanket temperature	2	B	SA	Delete
Safety rod manual (FH)	2	A	SA	Upgrade
Channels 1, 2, 3 period	2	A	DBD	Upgrade
Channels 9, 10, 11 level	2	A	DBD	Upgrade
Subassembly outlet temperature	2	A	DBD	Upgrade
Reactor outlet temperature	2	B	DBD	TBD
Coolant flow level	2	B	DBD	Upgrade
Bulk sodium level	2	C	SA	Delete
Control rod scram air assist pressure	2	C	SA	Delete
Control rods latched	2	C	SA	Delete

*A = Essential Function; B - Aid and Comfort Function; C - Nonessential Function; DBD = Design Basis Document; SA - Safety Analysis; TBD - To be determined.

TABLE III (cont.)

Protective Subsystem	Priority	Category*	Analysis Required	Proposed Action
Reactor plant gamma monitors	2	A	DBD	Upgrade
Channels 1, 2, 3 level	2	C	SA	Delete
Channels 1, 2, 3 high voltage	2	C	SA	Delete
Safety rod manual (RO)	2	C	SA	Delete
Bulk sodium temperature	2	B	DBD	TBD
Crane position interlock	2	C	SA	Delete
Nuclear instrument thimble temperatures	3	C	SA	Delete
Channel 7 (FH)	4	A	DBD	Upgrade
Primary pumps 1 and 2 (FH)	4	C	SA	Delete
Control rod position	4	C	SA	Delete
Safety rod position	4	C	SA	Delete
Channel 7 (RO)	4	C	SA	Delete
Upper plenum pressure	4	C	SA	Delete
Fuel handling complete	4	C	SA	Delete
2400-volt bus	4	C	SA	Delete
Auxiliary pump	4	C	SA	Delete
Control rods down	4	C	SA	Delete
Safety rods down	4	C	SA	Delete
Reactor building pressure	4	A	DBD	TBD

*A = Essential Function; B - Aid and Comfort Function; C - Nonessential Function; DBD = Design Basis Document; SA - Safety Analysis; TBD - To be Determined.

With the exception of the last two, these rationales were accepted as the basis for more detailed analysis in September 1971; and the deletions were subsequently made (see Table I).

As both the analytical and engineering effort progressed, it became apparent that the individual reactor protective subsystems under investigation could be grouped and related to the supporting analytical documents, with modifications processed by group. Table IV shows the groups and related safety documentation envisioned in May 1972.

Both analytical and engineering effort proceeded slowly along these general guidelines until late in 1973 when an accelerated program was initiated. Much of the supporting analysis was finalized in 1974, and the majority of the physical modifications were completed in 1975. The last of the planned trip deletions, reactor plant isolation, was performed in March 1976.

A summary of those modifications either completed or proposed as of February 1977 is given in Table I; supporting analysis is detailed in Table II. In addition, a safety evaluation for unrestricted fuel handling is in preparation that supports modifications to the trip circuit for fuel handling. Although Work Project 1011 was closed out in September 1976, the PPS upgrading effort continues as the need arises. The high plant factors achieved in 1975 and 1976 are due in part to the success of the PPS upgrading effort relative to its original intent--a reduction in the excessive number of spurious trips.

TABLE IV. POSTULATED GROUPING OF PPS PROTECTIVE SUBSYSTEMS FOR CONSIDERATION UNDER PPS UPGRADING EFFORT AS OF MAY, 1972

Protective Subsystem	Supporting Analysis	Subsequent Document No.	Subsequent PM No.
Earthquake detection system	DBD for earthquake protection	-	-
Reactor coolant flow Subassembly outlet temp. Reactor outlet temperature Primary pump trips Rate of change of flow 2400-V bus undervoltage	DBD for LOF Protection	ANL-75-31 (ANL/EBR-046)	443
Channels 1, 2, 3 period Channels 4, 5, 6 period Channel 7 Channels 9, 10, 11 level Channels 1, 2, 3, 4, 5, 9, 10, 11 high voltage Instrument thimble temp.[a] Control rod scram air assist pressure[b]	DBD for reactivity-related protection (RO)	ANL-75-32 (ANL/EBR-048)	753
Reactor building gamma[c] Reactor building pressure[c] Reactor building temp.[c] Subassembly outlet temp.[c] Reactor building isolation	DBD for reactor building containment	ANL/EBR-040 ANL-75-33 (ANL/EBR-072)	5088 5137
Control rod manual Safety rod manual Pump interlocks (FH) Crane position permissive Aux. pump permissive Control rods down permissive Safety rods up permissive Channels 1, 2, 3 level permissive	RDT Standard C16-1T	-	-

[a]Not analyzed in DBD for reactivity-related protection; deleted via PM 404.

[b]Not analyzed in DBD for reactivity-related protection.

[c]These are reactor building containment isolation functions.

TABLE IV (cont.)

Protective Subsystem	Supporting Analysis	Subsequent Document No.	Subsequent PM No.
Channels 1, 2, 3 period (FH)	Safety evaluation for unrestricted fuel handling	ANL-77-32	-
Channels 1, 2, 3 high voltage			
Channels 4, 5, 6 period (FH)			
Channel 7 (FH)			
Channels 4, 5, 6, 7 high voltage			
Control rod position (FH)			
Safety rod position (FH)			
FERD	Task 46 general analysis	ANL-76-94	5144
Bulk sodium level		ANL-75-34	5087
Bulk sodium temperature		ANL-75-34	5087
Argon blanket pressure	Task 46 general analysis	ANL-76-34	5069
Argon blanket temperature		ANL-76-34	5069
Fuel handling complete			431
Control rods not latched		ANL-76-34	405

IV. ANALYSIS OF SYSTEM AND DESCRIPTION OF MODIFICATIONS

A. Description of Original Reactor Shutdown System

The unmodified RSS contained the functions described in Table III and Fig. 35 of the HSR Addendum[2] and was as shown in Fig. 1. Known modifications to the RSS completed prior to the formalized PPS upgrading effort are noted individually in the sections describing each group of modifications.

B. Description of Present Reactor Shutdown System

The configuration of the modified RSS as of May 1977 is shown in Fig. 2. A detailed list of protective functions is found in Sect. 3.9.3 of the EBR-II Technical Specifications.

C. Detail of Modifications by Protective Subsystem Group

The grouping of Table IV will generally be followed in describing modifications performed and proposed.

1. Overall Protective Function

a. Description of Original System. The original shutdown system consisted of two reactor shutdown systems, one for reactor operation and one for unrestricted fuel handling, with the same parameters listed in Table III of Ref. 2.

b. Description of Modified System. A redundant shutdown system, designated System B, has been added in the reactor operate mode. The second shutdown system is connected to the original system (System A) in 1/2 trip logic and is composed of the following trips: power level, period, low flow, subassembly outlet temperature high, earthquake, safety rod manual, and control rod manual. Sensors and trip logic are identical to that used in System A (see Fig. 2).

The trip sensors are common to both Systems A and B. However, circuitry "downstream" of sensor input is independent for both systems.

No redundant shutdown system has been added in the fuel handling mode.

c. Safety Concerns Related to Protective Function. The event of concern is the possibility of failure of the reactor shutdown system circuitry associated with System A arising from a failure within the system itself.

d. Applicable Safety Analysis. The probability of the postulated failure of System A is highly remote. However, to comply with redundancy requirements as specified in RDT Standard C16-1T, it was recommended that the additional shutdown system be added. The trip parameters contained in System B are those identified as essential by safety analysis and referenced in EBR-II Technical Specifications. Considered are the overall shutdown system requirements, combining relevant considerations for each of the individual components.

e. Safety Conclusions. By providing redundancy in shutdown systems, overall reliability of the two systems has been improved. Included are all trip parameters shown to be essential for reactor operation.

f. Recommendations for System Upgrading. The addition of a second redundant shutdown system was recommended in order to comply with RDT Standard C16-1T.[5]

g. Summary of Modifications Performed or Proposed. A second shutdown system, isolated physically from the first, was installed and both systems connected to trip in a 1/2 trip logic by PM WAF-796.[5]

h. Compliance with C16-1T. The modified system meets the criteria of RDT Standard C16-1T.

2. Earthquake Detection System

a. Description of Original System. The original system for earthquake detection consisted of the following trips and sensors for both reactor operation and unrestricted fuel handling: Earthquake detection (1 detector, horizontal motion trip only).

b. Description of Modified System. The modified system for earthquake detection consists of the following trips and sensors for reactor operation (Systems A and B) and unrestricted fuel handling: Earthquake detection (3 detectors, 2/3 trip logic for both horizontal and vertical motion).*

c. Safety Concerns Related to Protective Function. The postulated events of concern related to earthquake are control rod binding and/or malfunction of containment isolation system subsequent to an earthquake.

d. Applicable Safety Analysis. Table VII of the HSR[1] does not list earthquake as a protective function; the only mention of earthquake in the HSR is on page 105, where it is stated: "Building construction is such as to prevent any anticipated earthquake activity from providing a source of difficulty." Table III of the HSR Addendum[2] lists the earthquake detection system as a trip function but provides no supporting analysis for its inclusion in the RSS.

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*Isolation trip capability is proposed for this system (see Sect. IV.C.5.b).

Although it was proposed that a design basis document for earthquake protection be prepared as part of the PPS upgrading effort (see Table IV), no formal seismic analysis to define structural response was ever performed for EBR-II. A general study was referenced in establishing Technical Specification setpoint limits and response times; this study assumed the probability of an earthquake of Magnitude IV on the Modified Mercalli scale. Seismic disturbances of this magnitude are described as being "felt indoors by many, outdoors by a few; at night, some awaken; dishes, windows, doors disturbed; motor cars rock noticeably." Although it was concluded that the EBR-II primary system could probably withstand tremors of greater magnitude, in lieu of more conclusive supporting analysis the trip point was established at a low level to ensure shutdown for minor tremors.

e. Safety Conclusions. Even though the probability of earthquake at the EBR-II site is small, protection is still considered necessary for this unlikely event. The earthquake trip has been retained and is shown in the EBR-II Technical Specifications as a required trip for both reactor operation and unrestricted fuel handling; trip settings, minimum response time, and minimum system configurations have been established as Limiting Conditions for Operation for reactor operation and unrestricted fuel handling. It has also been concluded that reactor building containment isolation upon earthquake should be added.

f. Recommendations for System Upgrading. The initial study by Boland (see Sect. III.B) recommends that two additional detectors be purchased, installed in diverse locations to avoid spurious trips due to system jarring, and connected to the RSS in 2/3 trip logic. Several component changes designed to improve system reliability have also been proposed over the years. And, as noted above, it is currently proposed as part of the effort to upgrade the containment isolation system that the earthquake detectors be added to the containment isolation trip circuit in a 2/3 trip logic (see Sect. IV.C.5).

g. Summary of Modifications Performed or Proposed. The original detector was replaced by three detectors that trip on both horizontal and vertical motion in 2/3 trip logic (PM 396). The three detectors are located in diverse areas; one in the sodium boiler building and two in the reactor building cable routing room tunnel. Circuit changes have been made that reduce the detectors' sensitivity to radio signals (Maintenance Change Notice-233(I)). The addition of isolation trip capability is pending approval of PM WAF-5137.

h. Compliance with RDT Standard C16-1T. The modified system is in full compliance with C16-1T.

3. Primary Flow Monitoring System

a. Description of Original System. The original system for primary flow monitoring consisted of the following trips and sensors for reactor operation:[2]

Low flow (5 flowmeters in a 1/5 trip logic, high and low pressure plenum inlet flow for both primary pumps and total outlet flow)

2 rate-of-change of flow (one for each primary pump)

Reactor outlet coolant temperature high (2 resistance thermometers)

Subassembly outlet temperature high (4 thermocouples in a 2/4 trip logic with both trip and isolation capability)*

10 primary pump trips (sensors for motor power off, winding temperature high, MG power off, MG clutch voltage low, MG cooling water pressure low, for both pumps, 1/1 trip logic for each sensor)

Auxiliary pump permissive interlock for startup (sensors for voltage low, current low, charge current high, discharge current high)

b. Description of Modified System. The modified system for primary flow monitoring consists of the following trips and sensors (System A) for reactor operation:

Low flow (3 flowmeters and 1 flow tube in a 1/4 trip logic; high pressure plenum inlet flow for No. 2 pump; low pressure plenum inlet flow for both pumps; total outlet flow)

Subassembly outlet temperature high (4 thermocouples in a 2/4 trip logic) with both trip and isolation capability)*

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*A discussion of the isolation capability of the trip on subassembly outlet temperature high is given in Sect. IV.C.5.

One primary pump trip (2400-V bus undervoltage, 1/1 trip logic.

Auxiliary pump permissive interlock for startup (sensors for voltage low, current low, charge current high)

Trips on low flow and subassembly outlet temperature high are also included in System B. Sensors and alarm functions have been retained for all deleted trips.

c. Safety Concerns Related to Protective Functions. Primary protection against fuel overtemperature as a result of power-to-flow mismatch during loss-of-flow (LOF) events is provided by trips on low flow; backup protection for LOF events is provided by a trip on subassembly outlet temperature high. (Trips on reactor outlet coolant temperature high were also included originally as backup to low-flow trips.) LOF events result in two temperature peaks. The first peak is determined by flow rate, reactivity feedback, trip rods reactivity worth, and trip circuitry setpoint and response time. Second peak temperatures are affected by trip rod reactivity worth, as well as fission product power, and auxiliary pump and/or convective flow rate. If the first temperature peak has occurred, the magnitude of the second peak is only weakly dependent on when the reactor is tripped. Because parameters related to RSS design (trip point and response time) are relevant only to the time of reactor trip, the safety analysis pertinent to the PPS upgrading thus directly involved first-peak temperature parameters only.

The specific LOF events of concern are loss of primary pumping power, with or without LOF trips; and single pump seizure, with or without simultaneous coastdown of the second pump or malfunction of flow trips. Other events identified were either judged to be too unlikely to be controlling or involved parameters affecting second peak temperatures only. The so-called "stuck rod" criterion, i.e., the assumption of sticking of a single control rod during trip, was taken into account when establishing current minimum limits on shutdown reactivity as defined in the EBR-II Technical Specifications.

d. Applicable Safety Analysis. Appendix A, Sect. 2, and Appendix F, Sect. 2, of the HSR[1] and Addendum,[2] respectively, report the original LOF safety analysis. Specifically, the following cases are considered: loss of all pumping power (including the auxiliary pump) with immediate and delayed trip; reactor trip followed by loss of all pumping power; loss of all pumping power without trip. Low flow, subassembly outlet temperature high, and primary pump power off were listed as trip parameters in the HSR (Table VII)[1] and Addendum (Table III).[2]

Section VIII-D of the EBR-II Status Report[3] reports on the following LOF events:

1. Loss of all pumping power without trip, no auxiliary or convective flow;
2. Loss of all pumping power with immediate trip, no auxiliary or convective flow;

3. Dual pump seizure without trip, no auxiliary or convective flow;
4. Dual pump seizure, trip within 2 s, no auxiliary or convective flow;
5. Single pump seizure with coastdown of second pump, without trip, 1.65% auxiliary flow, no convective flow; and
6. Single pump seizure with coastdown of second pump, trip in 0.05 s, 1.65% auxiliary flow, no convective flow.

In 1970, the HSR analysis of LOF events was reworked, using the then-existent PPS and improved computer techniques. Results of this analysis are reported in Ref. 6. At the same time, a preliminary evaluation of the system for primary flow monitoring (ANL/EBR-018, see Table II) concluded that the low-flow trips provided adequate protection against LOF events and the rate-of-change-of-flow and pump auxiliary function trips could be safely deleted from the RSS.

The design basis document for LOF protective functions,[7] in Sect. 3.1 through 3.5 and 6.5, identifies and addresses the following LOF faults, classified as anticipated or unlikely in accordance with C16-1T,[4] that were judged to define response time criteria and limiting setpoints for trips on low flow and subassembly outlet temperature high:

Loss of primary pumping power

Seizure of one pump

Seizure of one pump, coastdown of the second pump

Loss of primary pumping power and flow trip failure

Single pump seizure and failure of flow trips

All cases involving failure of flow trips assume trip due to subassembly outlet temperature high. Pertinent portions of the safety analysis of Ref. 7 are included in support of the plant modification upgrading LOF protection.[8]

Appendix B of Ref. 8 and Ref. 9 present the results of parametric studies investigating effects of changes of important variables upon RSS response during selected LOF events. The effects of reduced flow are addressed in Sect. VIII of Ref. 9. The operability requirement and minimum flow criterion for the auxiliary pump are also addressed in Ref. 9 (Sect. VII-E).

The Safety Analysis Statement for PM WAF-784 (see Table II) addresses the effectiveness of the trip on reactor outlet temperature high as backup for LOF events; the FSAA for PM WAF-5099[10] presents analysis in support of the replacement of a failing EM flowmeter with a ΔP flowmeter in the RSS. Both flowmeters measure total reactor flow at the outlet pipe of the reactor.

e. Safety Conclusions. Loss of flow protection by trips on both low flow and subassembly outlet temperature high is required for reactor operation to protect against fuel overtemperature upon LOF events. These requirements are so stated in the EBR-II Technical Specifications (limit 2.3.3); limiting safety system settings (LS^3), minimum response times, and minimum system configurations based on the analyses of Ref. 7 and 9 for trips on both low flow and subassembly outlet temperature high have been established for operation with both full and reduced flow. The limiting case for all LOF events considered is single pump seizure, with coastdown of the second pump.

It was further concluded that the anticipatory pump trips are not required as protective subsystems in the RSS.

The parametric analyses of Ref. 9 also verified limits on shutdown reactivity worth at power and maximum control rod worth and indicated the need for auxiliary pump operation, with 3.3% minimum flow capability for 3 min after trip, for reactor operations above 1 MWt. These requirements are so stated as Limiting Condiions for Operation in the EBR-II Technical Specifications.

Prior to this analysis, it was concluded that the trip on reactor outlet temperature high was both ineffective and redundant as backup to flow trips for a LOF event and not required as a protective function in the RSS.

Finally, it was concluded that the replacement ΔP flowmeter measuring total flow provides equal or better protection than the original flowmeter.

f. Recommendations for System Upgrading. The Boland study (see Sect. III.B) recommends: a reassessment of power and flow trip settings; deletion of the anticipatory pump failure trips and rate-of-change trips; a change to 2/5 trip logic for flow trips; deletion of the trip on reactor outlet temperature high; the replacement of subassembly-outlet-temperature trip thermocouples by others in the primary tank with a faster time response; and relocation of the auxiliary pump startup trip permissive interlock to the control circuit. The preliminary evaluation also recommended deletion of the trips on rate-of-change-of-flow and pump failure. The design basis document for LOF events established the need for trips on low flow and subassembly outlet temperature high only. System buffering, test source installation, and removal of anticipatory trips were recommended for compliance with C16-1T,[4] and the retention of the 2400-V bus undervoltage anticipatory pump trip was recommended as a mechanism to reduce first-peak temperatures upon loss of primary pumping power.[8] The replacement of total reactor flowmeter FM-514E with a Gentile flow tube (ΔP flowmeter), FE-541E, was recommended upon evidence of potential failure of FM-514E.

g. Summary of Modifications Performed or Proposed. Subsequent to publication of the HSR Addendum[2] and before July 1970, the following flow-related trips were added to the RSS:[3] 2400-V bus undervoltage and low pump current (both pumps). Also, the failure of the high-pressure-plenum inlet flow sensor for pump No. 1 necessitated the removal of the corresponding low flow trip and the flow rate-of-change trip for pump No. 1. Flow trip logic was changed to 1/4 at that time.

The trip on reactor outlet temperature high was deleted by PM WAF-784.

PM 443, Phases I and II, which was part of the PPS upgrading efforts, effected the following changes to the system:[8]

For the 4 low flow trips, installation of buffer amplifiers, millivolt test sources for channel calibration and associate interlock circuits; and mounting of components on new chassis.

For the 4 subassembly-outlet-temperature trips, installation of millivolt test sources for channel calibration and associated interlock circuits; and mounting of components on new chassis.

Deletion of anticipatory pump trips on both pumps for generator output breaker open or low current; high pump motor winding temperature; low clutch and brake cooling water pressure; low MG set clutch voltage; and MG set supply voltage breaker.

A millivolt test source for the 2400-V bus undervoltage trip was installed by PM 410.

Failed total reactor flowmeter FM-514E was replaced by flow tube FT-541E by PM WAF-5099.[10]

h. Compliance with C16-1T. The modified system for primary flow monitoring meets the requirements of C16-1T with the following variances, as noted in Sect. 4.5 of Ref. 8:

Section of C16-1T	Subject	Deviation
4.2.5	Fail-safe design (coincidence)	Not enough operable flowmeters to form coincidence circuit for pump No. 1
4.3.8	Independent of PPS wiring	Flow and subassembly outlet temperature sensor leads share a common conduit for a very short distance
3.5	Unnecessary Anticipatory Functions	Retention of 2400-V bus undervoltage trip

Further, the auxiliary pump permissive interlock for startup does not comply with Sect. 4.3.7, which precludes the use of protective functions to enforce administrative requirements. (This is, however, in compliance with Sect. 4.3.4, which specifies that signals from a system external to the PPS, i.e., control system, may be supplied to PPS actuator circuits provided with proper isolation.)

4. Neutron Monitoring System (Operate Mode)

a. Description of Original System. The original system for neutron monitoring consisted of the following trips and sensors for reactor operation:[2]

Source flux level low-permissive interlock for startup (3 log count rate low-range nuclear channels designated 1, 2, 3 in a 2/3 trip logic)

Period short (3 log count rate low-range nuclear channels designated 1, 2, and 3; and 3 log N intermediate range nuclear channels designated 4, 5, and 6; each in a 2/3 trip logic. Low-range period trip is bypassed at 400 W)

Linear level (one linear intermediate range nuclear channel designated as 7, normally bypassed for reactor operation)

Auto flux control rod down (one high-range nuclear channel designated 8 for automatic reactor control, bypassed for manual reactor operation)

Power level high--automatic and manual (3 linear high-range nuclear channels designated 9, 10, and 11, in 2/3 trip logic)

High voltage low channel trip (one voltage detector per nuclear channel; low voltage causes trip on affected channel).

b. Description of Modified System. The modified system for neutron monitoring consists of the following trips and sensors (System A) for reactor operation:

Source flux level low--permissive interlock for startup (Log Count Rate [LCR] subsystem of 3 wide-range nuclear channels designated A, B, and C in 2/3 trip logic)

Period short--bypassed above 30 MWt (LCR and Average Magnitude Squared [AMS] subsystems of wide-range channels A, B, and C in 2/3 trip logic)

Power level high (Linear Power Range [LPR] subsystem of wide-range nuclear channels A, B, and C in 2/3 trip logic)*

High voltage low channel trip (one voltage detector per nuclear channel; low voltage causes trip on affected channel)

Linear level (two linear intermediate range nuclear channels designated 7 and 7A, bypassed for reactor operation).

All but the channel 7 and 7A trips are included in System B.

c. Safety Concerns Related to Protective Function. Primary protection against fuel overtemperature as a result of transient overpower (TOP) events is provided by trips on period short and power level high; backup protection at higher power levels is provided by trips on subassembly outlet temperature high.

The specific TOP events of concern are reactivity increases due to rapid insertion of one or two control or safety rods.

d. Applicable Safety Analysis. Appendix A, Sect. 1, and Appendix F, Sect. 1, of the HSR[1] and HSR Addendum,[2] respectively, report the original TOP safety analysis. Specifically, the following cases applicable to the reactor operate mode are considered: both safety

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*Isolation trip capability is proposed for this trip (see Sect. IV.C.5.b).

rods, a central driver subassembly, or one control rod driven into a just-subcritical reactor; a single control rod driven in at full power and flow. The cases are considered both with and without protective trip functions. Period short and power level high are listed as trip parameters in the HSR (Table VII)[1] and Addendum (Table III).[2]

Section VIII-A of the EBR-II Status Report[3] provides an update of the original TOP analysis. In 1970, the HSR analysis of TOP events was again reworked, using the then-existent PPS and improved computer techniques.[6]

The design basis document for TOP protective functions,[11] in Sects. 3.0, 4.0, and 6.0, identifies and addresses the following TOP faults, classified as anticipated and unlikely in accordance with C16-1T,[4] that were judged to define response time criteria and limiting setpoints for power level and period trips:

Single control rod insertion (startup and full power)

Safety rod insertion (startup)

Simultaneous insertion of two control rods (startup and full power)

Pertinent portions of the safety analysis of Ref. 11 are included in support of the plant modification upgrading reactivity protection.[12]

Reference 13 presents the results of parametric studies investigating effects of changes of important variables upon RSS response during selected TOP events.

e. Safety Conclusions. Below 30 MWt, trip on short period is required to provide early protection against TOP events, with trip on power level high providing backup protection. Above 30 MWt, because of strong reactivity feedback, the period trip is ineffective and may be bypassed. There, trip on power level high provides primary protection against TOP events, with backup protection from trips on subassembly outlet temperature high. Accordingly, these requirements are so stated in the EBR-II Technical Specifications (limit 2.3.3); limiting safety system settings (LS)[3], minimum response times, and minimum system configurations based on the analysis of Refs. 11 and 13 for period and power level trips have been established for reactor operation. (Subassembly outlet temperature trip criteria are determined by LOF events--see Sect. IV.C.3.e).

f. Recommendations for System Upgrading. The Boland study (see Sect. III.B) recommends deletion of the automatic flux level trip from the RSS and reconnection of circuitry as a high flux alarm, bypass of period trip at 80% power, and possible deletion of the low-range period trip in the operate mode. The design basis document for TOP events reaffirms the need for power level and period trips only. System upgrading was recommended for compliance with C16-1T.

It has also been recommended that isolation trip capability be added to the trip on power level high to effect building isolation in the event of maximum hypothetical accident (MHA) (see Sect. IV.C.5.f).

g. Summary of Modifications Performed or Proposed. The automatic flux control system was never used. Consequently, prior to May 1970, the automatic flux control circuitry was deleted from the RSS[3] (PM 178C). Early in 1969, original nuclear channels 1, 2, 3, 7, 9, 10, and 11 were replaced by like channels to improve system reliability (PM 200) and linear channel 7A was added to the system as backup for channel 7 (PM 230). Sensor range for channels 1-6 was increased in December 1969 (PM 222). Log N channels 4, 5, and 6 were replaced by like channels to improve system reliability in 1972 (PM 398). Channels 1, 2, 3, 4, 5, 6, 9, 10, and 11 were replaced by wide-range nuclear channels A, B, and C by PM WAF-753 in 1975. PM WAF-753 also deleted the automatic power level trip from the RSS.

The addition of isolation capability to power level trips is pending approval and implementation of PM WAF-5137.

h. Compliance with C16-1T. The modified system for neutron monitoring meets the requirements of C16-1T with the following variance, as noted in Sect. 3.5 of Ref. 12:

Section of C16-1T	Subject	Variance
4.3.8	Independence of PPS wiring	Channel A and B preamplifier cables exit the reactor building through the same electrical penetration

Further, the source level count permissive interlock does not comply with Sect. 4.3.7, which disallows the use of protective functions to enforce administrative requirements. (It is, however, in compliance with Sect. 4.3.4, which specifies that signals from a system external to the PPS, i.e., control systems, may be supplied to PPS actuator circuits provided with proper isolation.)

5. Reactor Building Containment Isolation System

a. Description of Original System. The original system for reactor building containment isolation consisted of the following reactor and isolation system trips for both reactor operation and unrestricted fuel handling:[2]

Reactor building isolated, actuated by each of the following building isolation parameters:

Radiation level high (two gamma monitors, 1/2 trip logic, one bypassed for unrestricted fuel handling)

Reactor building air temperature high (3 temperature sensors, 1/3 trip logic)

Reactor building pressure high (1 pressure sensor, 1/1 trip logic)

Subassembly outlet temperature high (reactor operate only) (4 thermocouples, 2/4 trip logic).

b. Description of Modified System. The modified system for reactor building containment isolation does not result in reactor or fuel handling trip. The following trips initiate containment building isolation.

Radiation level high (two gamma monitors, 1/2 trip logic, one bypassed for unrestricted fuel handling).

Reactor building air temperature high (3 temperature sensors, 1/3 trip logic)

Reactor building pressure high (1 pressure sensor, 1/1 trip logic)

Subassembly outlet temperature high (4 thermocouples, 2/4 trip logic).

Pending approval and implementation of proposed PM WAF-5137, the system for reactor building isolation will be modified to include the following isolation trips.

Trips effecting partial building isolation (ventilation and purge lines only):

> Radiation level high (4 gamma monitors, located as follows: 2 on the reactor building ventilation exhaust and 2 on the air purge exhaust, 1/2 trip logic for each)

Trips effecting full building isolation (all lines exiting the reactor building):

> Earthquake (3 detectors that trip on both horizontal and vertical motion, 2/3 trip logic)
>
> Power level high (reactor operation only) or count rate high (unrestricted fuel handling only) (LCR or LPR sub-circuits of wide-range nuclear channels, 2/3 trip logic)
>
> Subassembly outlet temperature high (reactor operation only) (4 thermocouples, 2/4 trip logic; reactor operate mode only)

All lines with isolation valves may be isolated manually.

c. <u>Safety Concerns</u>. Reactor building isolation, with or without reactor trip, is required to assure containment integrity for events with the potential for significant radioactive release. Specific events of concern are the maximum hypothetical accident (MHA) and design basis accident (DBA),[1,2] radioactive sodium spillage, cover gas activity release, irradiated subassembly meltdown during fuel handling, and earthquake.

d. Applicable Safety Analysis. Table VII of the HSR does not list reactor building isolated as a protective function. Although trip on reactor building isolated is listed in Table III of the HSR Addendum,[2] no analysis is presented in support of its inclusion in the RSS. Appendix E, Sect. 2, of the HSR[1] and Appendix H of the HSR Addendum[2] describe the isolation system and indicate that the system sensors will initiate building isolation "in the...event of a significant nuclear incident or primary system sodium fire." Section IV-A of the HSR Addendum notes that "Trip of the isolation system will also scram the reactor."

A preliminary analysis of the isolation system (ANL/EBR-040--see Table II) addresses the potential sources of events requiring isolation and makes a number of recommendations for system upgrading. The original basis document for containment isolation, ANL/EBR-072, (see Table II), in Sect. IV, addresses the following events:

Activity from sodium fires

Activity from cover gas release

Activity from fuel element handling system

Activity from MHA

Plutonium release

Pertinent portions of this safety analysis are included in support of the plant modification proposing deletion of the trip on reactor building isolated from the RSS.[14]

A revised basis document addresses the same events plus earthquake.[15] Pertinent portions of this safety analysis have been included in support of the proposed modification to upgrade the building isolation system.[16]

e. Safety Conclusions. It was concluded that trip on reactor building isolated is not required in the RSS. None of the identified events requiring building isolation can be mitigated in any way by automatic reactor trip on building isolation. Further, it was concluded that (1) reactor building pressure and temperature high are ineffective isolation parameters for sodium fire or the MHA; and (2) the earthquake detection, power level high, count rate high, and subassembly outlet temperature high are effective isolation parameters for earthquake, the MHA, and certain flow blockage events, respectively.

Required plant conditions, minimum configurations, and response times have been defined for the isolation system and isolation trip parameters and are designated Limiting Conditions for Operation in the EBR-II Technical Specifications for reactor operation and fuel handling (both restricted and unrestricted).

f. Recommendations for System Upgrading. The Boland study (see Sect. III.B) recommends removal of the trip on reactor building isolation contingent upon satisfactory results from heatup tests with the reactor building isolated. The recommendation and rationale for deletion of this trip was presented to and accepted, pending further analysis, by ERDA (then AEC) as part of the initial PPS upgrading effort (see Sect. III.B).

The preliminary safety analysis in ANL/EBR-040 (see Table II) recommends the removal of the isolation trip on subassembly outlet temperature high; relocation of existing radiation monitors and addition of four more, making two sets of three monitors each with 2/3 trip logic; deletion of the isolation trip on reactor building temperature high; and addition of two pressure monitors, making a system with 2/3 trip logic.

Subsequent detailed safety analysis reversed some of these earlier recommendations and has been used as the basis for currently proposed modifications. Specifically, the original design criteria document for containment isolation, ANL/EBR-072 (see Table II), recommends the deletion of the trip on reactor building isolated from the RSS; deletion of trips on reactor building temperature and pressure high; addition of an isolation trip on earthquake; the retention of the isolation trip on subassembly outlet temperature high; installation of three gamma monitors on the building ventilation exhaust stack with isolation trip capability in 2/3 trip logic; and the deletion of the two original gamma monitors from the isolation circuit. The revised basis document for containment isolation[15] further recommends: two automatic isolation modes, designated as partial and full; the addition of earthquake, power level and count rate high, and the retention of subassembly outlet temperature high as full isolation trips; and the relocation of the two gamma monitors to the building ventilation exhaust line and addition of two more on the building purge exhaust line, each as partial isolation trip sensors in 1/2 trip logic.

g. Summary of Modifications Performed or Proposed. Prior to 1970, a manual partial isolation of ventilation lines was added to the system (no PM number). The trip on reactor building isolated was deleted from the RSS by PM WAF-5088. System alarm functions and isolation trip capability were retained with the original trip logic. The recommendations of Ref. 15 have been proposed as part of PM WAF-5137, currently under review.

h. Compliance with RDT Standard C16-1T. The deletion of the reactor building isolated trip is in compliance with Sect. 3.5 of C16-1T which requires the removal of unnecessary anticipatory trips.

6. Instrument Thimble Temperature Monitoring

a. Description of Original System. The original system for instrument thimble temperature monitoring consisted of the following trips and sensors for both reactor operation and unrestricted fuel handling:[2]

> Instrument thimble detector mounting, Bank A, temperature high (4 thermocouples, 2/4 trip logic)
>
> Instrument thimble detector mounting, Bank B, temperature high (4 thermocouples, 2/4 trip logic)

b. Description of Modified System. The modified system for instrument thimble temperature monitoring consists of the following parameters and circuit logic providing an alarm function only for reactor operation and unrestricted fuel handling:

> Instrument thimble temperature high (6 thermocouples, 2/6 trip logic)

c. Safety Concerns Related to Protective Function. The postulated events of concern relative to thimble overtemperature are loss of thimble cooling or fire resulting from sodium in-leakage.

d. Applicable Safety Analysis. Table VII of the HSR[1] does not list trip on thimble temperature high as a protective function. Although trip on thimble temperature high (Banks A and B) is listed in Table III of the HSR Addendum,[2] no analysis is presented in support of its inclusion in the RSS.

The engineering submittal in support of the removal of the trip on instrument thimble temperature high provides the results of thimble heatup tests simulating loss of cooling events (see Table II).

e. Safety Conclusions. It was concluded that trip on instrument thimble temperature high is not required in the RSS. The thimble heatup rates upon either loss of flow or fire are slow enough to allow ample time for operator action prior to heat damage to cabling or sensors.

Required plant conditions and minimum configurations have been defined for instrument thimble temperature monitoring and are designated Limiting Conditions for Operation in the EBR-II Technical Specifications for reactor operation and unrestricted fuel handling.

f. Recommendations for System Upgrading. The Boland study (see Sect. III.B) recommends the deletion of the trip on thimble temperature high with retention of alarm capability. The recommendation and rationale for deletion of this trip was presented to and accepted by ERDA (then AEC) as part of the initial PPS upgrading effort (see Sect. III.B).

g. Summary of Modifcations Performed or Proposed. Subsequent to publication of the HSR Addendum and May 1970, the system was modified to include six Bank A thermocouples as trip sensors, in a 2/6 trip logic. (Bank B thermocouples provided an alarm function only.) The trip on instrument thimble temperature high was deleted from the RSS by PM 405. System alarm functions were retained with the same circuit logic as for the trips, thus maintaining two redundant alarm systems.

h. Compliance with RDT Standard C16-1T. The deletion of this trip is in compliance with Sect. 3.5 of C16-1T which requires the removal of unnecessary anticipatory trips.

7. Control Rods Latched Monitoring

a. Description of Original System. The original system for control rods latched monitoring consisted of the following trips and sensors for reactor operation:[2]

Any control rod unlatched (2 latch switches per control rod, either switch open would activate trip).

b. Description of Modified System. The modified system for control rods latched monitoring consists of the same sensors and circuit logic functioning as a permissive interlock for reactor startup and providing an alarm function only during reactor operation. Circuitry to control rod positions not occupied by control rods has been deactivated.

c. Safety Concerns Related to Protective Function. The postulated events of concern relative to trip for any control rod unlatched is the dropping of one or more control rods out of core, resulting in flux distortion and power loss. Control rod drive binding could also cause the unlatched condition.

d. Applicable Safety Analysis. Table VII of the HSR[1] does not list a trip for any control rod unlatched, although in Sect. V-A it is stated that "The control rod drive mechanism is such as to prevent a rod from...failing to unlatch without knowledge of the operator." Although trip for any control rod unlatched is listed in Table III of the HSR Addendum,[2] no analysis is presented in support of its inclusion in the RSS.

Analysis provided in support of the removal of the trip on any control rod unlatched[17] identifies postulated faults of concern.

e. Safety Conclusions. It was concluded that a trip on any control rod unlatched is not required in the RSS during reactor operation. Flux distortion resulting from a dropped rod (or rods) was not considered to be a safety problem, nor was control rod drive binding. A significant power loss would require operator adjustment of secondary sodium flow to prevent overcooling the primary sodium, but, again, this was not judged a safety concern. The postulated reason for inclusion of the trip on any control rod unlatched in the RSS is for reactor operation with the automatic flux control system; this system was never brought into service.

It was concluded, however, that assurance that all control rods are latched should be provided before reactor startup. Required plant conditions and minimum configurations have been defined for control rod latch monitoring and are designated Limiting Conditions for Operation in the EBR-II Technical Specifications for reactor startup.

f. Recommendations for System Upgrading. The Boland study (see Sect. III.B) recommends the deletion of the trip on any control rod unlatched with retention of alarm capability. The recommendation and rationale for deletion of this trip was presented to and accepted by ERDA (then AEC) as part of the initial PPS upgrading effort (see Sect. III.B).

g. Summary of Modifications Performed or Proposed. The trip on any control rod unlatched for reactor operation was converted to a permissive interlock for startup by PM 404. System alarm functions were retained with the same circuit logic as for the trip, and a test switch for interlock checks of the limit switches was provided.

The circuitry to control rod positions converted to test facilities (INCOT's or INSAT's) was deactivated as part of the conversion modifications.

h. Compliance with RDT Standard C16-1T. The deletion of this trip is in compliance with Sect. 3.5 of C16-1T which requires the removal of unnecessary anticipatory trips. However, its retention as a permissive interlock for startup does not comply with Sect. 4.3.7, which precludes the use of protective functions to enforce administrative requirements. (It is, however, in compliance with Sect. 4.3.4, which specifies that signals from a system external to the PPS, i.e., control systems, may be supplied to PPS actuator circuits provided with proper isolation.)

8. Control Rod Air Assist Pressure Monitoring

a. Description of Original System. The original system for control rod air assist pressure monitoring consisted of the following trips and sensors for reactor operation:[2]

Control rod scram assist air pressure low (1 pressure monitor, 1/1 trip logic)

b. Description of Modified System. The original system for control rod air assist pressure monitoring has not been modified.

c. Safety Concerns Related to Protective Function. The postulated event of concern is a reduction of control rod drop time due to total loss of air assist pressure which could, in turn, affect peak fuel temperatures during TOP events.

d. Applicable Safety Analysis. Although Table VII of the HSR[1] does not list a trip on control rod air assist pressure low, Section III.A.6.a on control drive systems states that "...Pressure-actuated switches scram the reactor in the event of failure of the air supply." Trip on control rod air assist pressure low is listed in Table III of the HSR Addendum.[2] However, no analysis in support of its inclusion in the RSS is presented in either document.

Section VII of Ref. 13 presents the effect of increasing control rod drop time on peak temperatures during TOP events.

e. Safety Conclusions. It was concluded that a trip on control rod air assist pressure low is not required in the RSS. Air assist is not required to keep control rod drop times within those bounds assuring that peak temperatures associated with TOP events will not exceed safety limits. Accordingly, the EBR-II Technical Specification limit on control rod drop time is stated without air assist, and trip on loss of air assist pressure is not identified as essential. Required plant conditions and minimum configurations have been defined for control rod air assist pressure and are designated Limiting Conditions for Operation in the EBR-II Technical Specifications for reactor operation.

f. Recommendations for System Upgrading. The Boland study (see Sect. III.B) recommends deletion of the trip on control rod air assist pressure low if it can be shown that lengthened drop times will not affect the results of HSR accident analysis. The recommendation and rationale for deletion of the trip on control rod air assist pressure low was presented to ERDA (then AEC) as part of the initial PPS upgrading effort but was deferred at that time pending the determination of the essential performance requirements for control rod trip mechanisms.

g. Summary of Modifications Performed or Proposed. Although the supporting analysis exists, no proposal to delete the trip on control rod air assist pressure low from the RSS is planned at this time.

h. Compliance with C16-1T. The retention of this trip does not comply with Sect. 3.5 of C16-1T which requires the removal of unnecessary anticipatory trips. Further, the 1/1 trip logic does not comply with Sect. 4.2.1 and 4.2.2 on internal random failures and redundancy.

9. Safety Rods Up Monitor (Operate Mode)

a. Description of Original System. The original system for safety rods up monitoring consisted of the following interlock trips and sensors for reactor operation:[2]

Safety rods not in "full up" position (2 limit switches, 1/2 trip logic)

b. Description of Modified System. The basic function of the system for safety rods up monitoring has not been modified.

c. Safety Concerns Related to Protective Function. The postulated event of concern during reactor operation relative to safety rods up is the TOP event associated with the inadvertent insertion of safety rods into a critical core.

d. Applicable Safety Analysis. The TOP safety analysis pertinent to reactor operation may be found in Ref. 1, 2, 3, 6, 11, and 12 (see Sect. IV.C.4.c).

Although trip on safety rods not up is not listed in Table VII of the HSR,[1] there are several references in the text to the fact that reactor operation may not be performed with the safety rods down or partially inserted, even though no credit is taken for safety rod worth in reactivity control. For example, in Sect. III.A.6 of the HSR, it is stated that "Two safety rods are provided in the reactor in addition to the 12 operational control rods. The safety rods are not a part of the normal operational control system for the reactor. The safety rods are always in the reactor and they are designed to function when the control rods are disconnected from their drives..." Section III.A.6.b states "...All reactor operations, including actuation of the control system...require the safety rods to be in the up position."

Although trip on safety rods not fully up is listed in Table III of the HSR Addendum,[2] no analysis is presented in support of its inclusion in the RSS.

e. Safety Conclusions. It has been concluded that the trip on safety rods not fully up is not required in the RSS and that reactor operation may be safely performed with the safety rods only partially inserted. Although the EBR-II Technical Specifications require that the safety rods provide shutdown reactivity $\geq$ 120% of the power reactivity decrement, credit is not taken for safety rod action for events requiring automatic trip. Limits applied to control rods ensure that sufficient shutdown reactivity is available to accommodate all identified faults, including insertion of both safety rods with the reactor at power.

Because analysis in support of reactor operation with safety rods partially inserted has not been formalized, the EBR-II Technical Specifications currently require that reactor operation be conducted with safety rods up. Should a proposal be reinstated to delete the safety rods up trip from the RSS, the appropriate limiting conditions for operation will be developed for reactor operation with safety rods down.

f. Recommendations for System Upgrading. The Boland study (see Sect. III.B) recommends the deletion of the trip on safety rods not fully up from the RSS for reactor operation, but no further action was taken at that time. Removal of the trip on safety rods not fully up from the RSS (reactor operate mode) was later proposed as part of PM 433, a modification to use safety rods as reactivity shims during reactor operation. PM 433 was subsequently cancelled, and no proposal to delete the trip on safety rods not up from the RSS is planned at this time.

g. Summary of Modifications Performed or Proposed. To avoid excessive spurious trips during the initial and final stages of unrestricted fuel handling when reactivity insertion events are not possible, a safety rods up bypass was installed by PM 226. A similar bypass for use during rotating plug seal cleaning was installed by PM 338.

No proposal to delete the trip on safety rods not up from the RSS is planned at this time.

h. Compliance with C16-1T. The retention of this trip does not comply with C16-1T which disallows the retention of unnecessary trips.

10. Bulk Sodium Temperature and Level Monitoring

a. Description of Original System. The original system for bulk sodium temperature and level monitoring consisted of the following trips and sensors[2] for reactor operation:

Bulk sodium temperature high (4 thermocouples in 2/4 trip logic)

Bulk sodium level high (1 level transducer, 1/1 trip logic)

Bulk sodium level low (1 level transducer, 1/1 trip logic)

b. Description of Modified System. The modified system for bulk sodium temperature and level monitoring consists of the same parameters and circuit logic providing an alarm function only for reactor operation. The original sensors have been replaced.

c. Safety Concerns Related to Protective Function. The postulated events of concern relative to bulk sodium level high are (1) a rupture of secondary sodium inlet piping to the intermediate heat exchanger (IHX), which would result in partial drainage of secondary sodium into the primary tank, and (2) the leakage of shutdown coolers.

The postulated events of concern relative to bulk sodium level low are (1) drainage of sodium from the primary tank due to primary tank failure, or (2) drainage due to leakage in the fuel element rupture detector (FERD) loop or primary purification system. These events, in turn, could have the potential for dropping the sodium level below primary pump inlets.

The postulated event of concern for bulk sodium temperature high is the loss of secondary sodium system cooling capacity.

d. Applicable Safety Analysis. Although trips on bulk sodium level and temperature high and low are listed in Table VII of the HSR,[1] no analysis is presented in support of their inclusion in the RSS. In fact, the only mention of bulk sodium level and temperature in the HSR (pp. 12 and 56) infers that trips on low level and high temperature are not required. The RSS defined by Table III of the HSR Addendum[2] includes only bulk sodium level high and low and temperature high as trip parameters for reactor operation, again with no supporting analysis.

Fault tree analysis in support of the removal of the trips on bulk sodium level high and low and temperature high[18] identifies postulated events of concern, and detailed analyses are provided for those events for which additional analysis was indicated; i.e., IHX rupture, primary tank failure, and loss of secondary sodium cooling capability.

e. Safety Conclusions. It was concluded that trips on bulk sodium level low and high and temperature high are not required in the RSS. Bulk sodium level rise due to IHX failure is insufficient to contact the reactor vessel cover; bulk sodium temperature rise due to loss of secondary sodium cooling can be accommodated within the stress capacity of the primary tank. The drop in sodium level due to rupture of the primary tank inner vessel is not sufficient to uncover the primary pumps; a rupture of both inner and outer vessels could result only from the DBA, for which trip on bulk sodium level low is not effective.

Required plant conditions and minimum configurations have been defined for bulk sodium level and temperature and are designated Limiting Conditions for Operation in the EBR-II Technical Specifications in both the reactor operate and unrestricted fuel handling modes.

f. Recommendations for System Upgrading. The Boland study (see Sect. III.B) recommends the deletion of trips on bulk sodium level low and high from the RSS with retention of alarm capability for the low level condition. In the case of bulk sodium temperature high, Boland indicates the trip is probably not warranted but did not recommend its deletion.

The recommendation and rationale for deletion of trips on bulk sodium level high and low was presented to and accepted by ERDA (then AEC) as part of the initial PPS upgrading effort (see Sect. III.B). The deletion of the trip on bulk sodium temperature high was recommended as part of this effort at a later date.

g. Summary of Modifications Performed or Proposed. The trips on bulk sodium level high and low and temperature high were deleted from the RSS by PM WAF-5087. System alarm functions were retained with the same circuit logic as for the trips. Prior to the deletion of these trips from the RSS, the bulk sodium level sensor and bulk sodium temperature thermocouples were replaced to enhance system reliability (PM 231, PM 264).

h. Compliance with RDT Standard C16-1T. The deletion of these trips is in compliance with Sect. 3.5 of C16-1T which requires the removal of unnecessary anticipatory trips.

11. Argon Cover Gas Temperature and Pressure Monitoring

a. Description of Original System. The original system for argon cover gas temperature and pressure monitoring consisted of the following trips and sensors[2] for both reactor operation and unrestricted fuel handling:

Argon blanket temperature high (thermal element, 1/1 trip logic)

Argon blanket pressure high (pressure transmitter, 1/1 trip logic)

b. Description of Modified System. The modified system for argon cover gas temperature and pressure monitoring consists of the same sensors and circuit logic providing an alarm function only for both reactor operation and unrestricted fuel handling.

c. Safety Concerns Related to Protective Function. The postulated event of concern relative to argon cover gas temperature high is sodium fire in the primary tank, resulting from air ingress due to (1) primary tank rupture, (2) rupture of the fuel unloading machine (FUM) blower inlet line, (3) cover gas system failure, or (4) rotating plug failure.

The postulated events of concern relative to argon cover gas pressure high are (1) failure of the FUM argon system and (2) rupture of the IHX inlet piping.

d. Applicable Safety Analysis. Although a trip on argon cover gas pressure high is listed in Table VII of the HSR,[1] no analysis is presented in support of its inclusion in the RSS (p. 65). Argon cover gas temperature high is listed as an alarm function only, as an indicator of sodium fire (p. 66). The RSS defined by Table III of the HSR Addendum[2] includes both argon blanket temperature and pressure high as trip parameters for reactor operation and unrestricted fuel handling with no supporting analysis.

Fault tree analysis in support of the removal of trips on argon cover gas temperature and pressure high[19] identifies postulated events of concern; and detailed analyses are provided for those events for which additional analysis is indicated, i.e., primary tank rupture, FUM blower line rupture, and IHX inlet line rupture.

e. Safety Conclusions. It was concluded that trips on argon cover gas temperature and pressure high were not required in the RSS. The likelihood of either temperature or pressure sensors detecting a sodium fire resulting from air inleakage through the FUM was shown to be highly remote, and in no event would temperature or pressure reach the trip setpoints. Even assuming the complete failure of both primary tank pressure relief mechanisms (vacuum-pressure relief system and floating head tank), peak pressure resulting from the IHX rupture incident would not exceed allowable primary tank stress limitations. In neither case would reactor trip have any effect on the outcome of the incident. Primary tank rupture could result only from the MHA, for which trips on argon cover gas pressure or temperature high were not effective.

Required plant conditions and minimum configurations have been defined for argon cover gas pressure and temperature and are designated Limiting Conditions for Operation in the EBR-II Technical Specifications for reactor operation and fuel handling (both unrestricted and restricted).

f. Recommendations for System Upgrading. The Boland study (see Sect. III.B) recommends the deletion of trips on argon cover gas temperature and pressure high from the RSS with retention of high level alarm capability. The recommendation and rationale for deletion of these trips was presented to and accepted, pending further analysis, by ERDA (then AEC) as part of the initial PPS upgrading effort (see Sect. III.B).

g. Summary of Modifications Performed or Proposed. The trips on argon cover gas temperature and pressure high were deleted from the RSS by PM WAF-5069. System alarm functions were retained with the same circuit logic as for the trips.

h. Compliance with RDT Standard C16-1T. The deletion of these trips is in compliance with Sect. 3.5 of C16-1T which requires the removal of unnecessary anticipatory trips.

12. Delayed Neutron Detection

a. Description of Original System. The original system for delayed neutron detection consisted of the following trips and sensors for reactor operation and was not part of the system described in the HSR Addendum:

> Fuel element rupture detection (FERD) loop count level high (3 delayed neutron detector channels, 2/3 trip logic)
>
> Fuel element rupture detection (FERD) high voltage low (1 voltage monitor per channel, 1/1 trip on each channel)

b. Description of Modified System. The modified system for delayed neutron detection consists of the same sensors and circuit logic for reactor operation, providing a detection and alarm function only.

c. Safety Concerns Related to Protective Function. The postulated events of concern relative to delayed neutron detection capability are events resulting in potentially propagative fuel-cladding breach accompanied by loss of bond sodium. These events may be categorized as (1) whole-core events, such as unprotected LOF or TOP events leading to core meltdown; (2) single subassembly events, such as flow blockage leading to meltdown; and (3) single element events, or breaches.

d. Applicable Safety Analysis. The FERD system was installed both as a detection device and protective function in the RSS sometime in the late 1960's. The first formal mention of the FERD in an EBR-II safety document is in Fig. 43 of Ref. 3, an updated version of Fig. 35 of the HSR Addendum[2] depicting the RSS configuration. No analysis is presented in support of its addition as a trip function; in fact, no mention is made of the system at all in the text of Ref. 3.

The safety evaluation in support of FERD system upgrading[20] presents detailed analysis of the one event identified as pertinent to the FERD trip: flow blockage to a single EBR-II driver fuel subassembly with postulated propagation of failure to adjacent subassemblies. This analysis is summarized in supporting documentation for the proposal to remove the FERD trip from the RSS.[21]

e. Safety Conclusions. It was concluded that trips on FERD loop count level high and FERD high voltage low were not required in the RSS. System delay time eliminates the FERD trip as an effective protective device for whole-core events. Time delays also render the FERD trip ineffective against nonpropagative whole subassembly failures. Rapid single element propagative events are discounted based on the results of fast reactor operating experience. Slowly propagating whole subassembly failures, judged to be highly unlikely, would proceed slowly enough to allow ample time for manual trip upon receipt of FERD system alarm.

Required plant conditions and minimum configurations have been defined for the FERD system and are designated Limiting Conditions for Operation in the EBR-II Technical Specifications for reactor operation.

f. Recommendations for System Upgrading. The Boland study (see Sect. III.B) infers that the addition of the FERD trip to the RSS was an internal administrative decision and suggests that its removal could be accomplished in the same manner. The recommendation and rationale for deletion of the FERD trip was presented to ERDA (then AEC) as part of the initial PPS upgrading effort but was deferred at that time due to the lack of comprehensive analysis. The safety evaluation in support of upgrading the FERD system[20] was ultimately prepared and recommends deletion of the FERD as a protective subsystem in the RSS and the upgrading of FERD system sensitivity to improve its capability for delayed neutron detection.

g. Summary of Modifications Performed or Proposed. The trips on FERD high count level and low voltage were deleted from the RSS by PM WAF-5144. System alarm functions were retained with the same circuit logic as for the trip. System improvements to upgrade and increase system sensitivity will be effected by PM WAF-5168.

h. Compliance with RDT Standard C16-1T. The deletion of this trip is in compliance with Sect. 3.5 of C16-1T which requires the removal of unnecessary anticipatory trips.

13. Reactor Outlet Plenum Pressure Monitoring

a. Description of Original System. The original system for reactor outlet plenum pressure monitoring consisted of the following trips and sensors for reactor operation:[2]

> Reactor outlet plenum pressure high (1 pressure sensor, 1/1 trip logic)

b. Description of Modified System. The basic function of the original system for reactor outlet plenum pressure monitoring has not been modified.

c. Safety Concerns related to Protective Function. The postulated event of concern relative to reactor outlet plenum pressure high is outlet flow blockage, with the resultant pressure increase causing the reactor vessel cover to lift.

d. Applicable Safety Analysis. Although trip on reactor outlet plenum pressure high is listed in both Table VII of the HSR[1] and Table III of the HSR Addendum,[2] no analysis is presented in either document in support of its inclusion in the RSS. Section IV.A.3.a(2) of the HSR[1] merely states that "...The 3 plenum pressures are continuously recorded, the recorders being provided with high and low pressure contacts for...scram in the case of the outlet plenum."

Section VI.B.1.a of ANL/EBR-062 (see Table II) addresses the probability of cover lift due to an overpressure in the upper plenum.

e. Safety Conclusions. It was concluded that trip on reactor outlet plenum pressure high is not required in the RSS. The postulated event is in the extremely unlikely category defined by RDT Standard C16-1T and thus is outside the scope of RSS protection. Further, decrease in flow from any cause will result in reactor trip from low flow or subassembly outlet temperature high.

Accordingly, the EBR-II Technical Specifications do not require reactor trip on plenum overpressure. Required plant conditions and minimum configurations have been defined for reactor outlet plenum pressure and are designated Limiting Conditions for Operation in the EBR-II Technical Specifications for reactor operation.

f. Recommendations for System Upgrading. The Boland study (see Sect. III.B) recommends deletion of the trip on reactor outlet plenum pressure high from the RSS but no further action has been taken.

g. Summary of Modifications Performed or Proposed. No proposal to delete the trip on reactor outlet plenum pressure high is planned at this time.

h. <u>Compliance with C16-1T</u>. The retention of this trip does not comply with Sect. 3.5 of C16-1T which requires the removal of unnecessary anticipatory trips. Further, the 1/1 trip logic does not comply with Sect. 4.2.1 and 4.2.2 on internal random failures and redundancy.

14. Fuel Handling Complete Monitoring

a. Description of Original System. The original system for fuel handling complete monitoring consisted of the following trips and circuit conditions for reactor operation:[2]

Fuel handling operation incomplete (2 latches unlatched and one key switch off, 1/3 trip logic)

Reactor vessel cover not completely down (cover down, 1/1 trip logic)

Reactor vessel cover: any lock not in locked position (3 cover locks locked, 1/3 trip logic)

Reactor vessel cover: any lock holddown force low (3 cover holddown springs compressed, 1/3 trip logic)

b. Description of Modified System. The modified system for fuel handling complete monitoring consists of the following permissive interlock circuits for reactor startup:

Fuel handling operation incomplete, consisting of the following parameters:

Sequence A and H complete (2 latches unlatched, 1/2 circuit logic)

Control power (KS-2) off (1 key switch off, 1/1 circuit logic)

Cable connectors (15 connectors connected, 1/15 circuit logic)

INCOT-I-III (3 subassembly down and yoke engaged sensors, 1/6 circuit logic)

Reactor vessel cover (cover down, 1/1 circuit logic)

Reactor vessel cover locks (3 cover locks locked, 1/3 circuit logic)

Reactor vessel cover holddown spring (3 torque switches closed, 1/3 trip logic)

In addition, the latter three functions have been retained as trip functions in the RSS for reactor operation.

c. Safety Concerns Related to Protective Function. The postulated events of concern relative to fuel handling complete are control rod connector failures and control rod binding due to the reactor cover lifting or tilting.

d. Applicable Safety Analysis. Table VII of the HSR[1] lists a trip on reactor cover unlatched but provides no analysis in support of its inclusion in the RSS. Although fuel handling operation incomplete and the 3 cover trips are listed as trip functions for reactor operation in Table III of the HSR Addendum,[2] no analysis is presented in support of their inclusion in the RSS. Rather, Sect. IV.J of the HSR Addendum[2] lists the fuel handling complete circuit conditions as conditions required to satisfy only a permissive interlock for reactor startup.

ANL/EBR-062, the document that provides an evaluation of the fuel handling complete circuit in support of its conversion to a startup permissive (see Table II), identifies postulated faults of concern.

e. Safety Conclusions. It was concluded that trip on fuel handling not complete is not required in the RSS. Connectors are either fail-safe, or postulated failures lead to conditions within the defined bounds of TOP analysis.[11] Events leading to reactor cover lifting and/or tilting are defined as extremely unlikely[4] and thus beyond the scope of RSS protection.

However, because of the potential for control rod binding in this extremely unlikely event, the EBR-II Technical Specifications do require reactor trip on evidence of cover lifting or unlocked. Required plant conditions and minimum configurations have been defined for fuel handling complete and the reactor vessel cover and are designated Limiting Conditions for Operation in the EBR-II Technical Specifications for reactor startup and operation.

f. Recommendations for System Upgrading. The Boland study (see Sect. III.B) withholds judgment on removal of this trip pending analysis of potential for the cover lifting. The March 1972 version of ANL/EBR-062 (see Table II) recommends that all fuel handling complete circuit conditions be retained in a permissive interlock circuit for reactor startup. Even though not required from a safety standpoint, the revised version of February 1974 further recommends that the three reactor vessel cover trips be retained during reactor operation.

g. Summary of Modifications Performed or Proposed. The cable connector and instrumented subassembly circuits were added some-time prior to 1970 (no PM no.). The system for fuel handling complete monitoring was converted to a permissive interlock circuit for reactor startup by PM 431, with reactor vessel cover trips retained on a separate circuit for reactor operation. Cabling and 100-pin connectors for the 3 control rod positions converted to in-core test facilities were removed from the cable connector subsystem by PM WAF-5130.

h. Compliance with RDT Standard C16-1T. The modified system for fuel handling complete monitoring adheres to the defined criteria of C16-1T with the following variance, as noted in Sect. IV.C of ANL/EBR-062 (February 1974):

> Although it was concluded that the reactor vessel cover trips are not required, they were retained in the RSS and not upgraded in accordance with the criteria of C16-1T.

Further, the retention of this system as a permissive interlock for startup does not comply with Sect. 4.3.7, which disallows the use of protective functions to enforce administrative requirements. (It does, however, comply with Sect. 4.3.4, which specifies that signals from a system external to the PPS, i.e., control systems, may be supplied to PPS actuator circuits provided with proper isolation.)

15. Permissive Interlocks for Reactor Startup

a. Description of Original System. The original system contained the following permissive interlock circuits for reactor startup:[2]

Source flux level low (3 log count rate low-range nuclear channels, designated 1, 2, and 3, 1/3 circuit logic)

Any control rod not in "full down" position (12 sensors, 1/12 circuit logic)

Auxiliary pump rectifier output voltage low (1 voltage sensor, 1/1 circuit logic)

Auxiliary pump input current low (1 current sensor, 1/1 circuit logic)

Auxiliary pump battery not fully charged (charge current high) (1 current sensor, 1/1 circuit logic)

b. Description of Modified System. The modified system contains the following permissive interlock circuits for reactor startup:

Source flux level low--see Sect. IV.C.4*

Any control rod not down (9 limit switches, 1/9 circuit logic)

Auxiliary pump operating--see Sect. IV.C.3*

Crane position satisfactory (2 limit switches, 1/2 circuit logic)

Fuel handling incomplete--see Sect. IV.C.14*

Any control rod not latched--see Sect. IV.C.7*

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*These functions addressed in other sections will not be discussed further in this section.

c. Safety Concerns Related to Protective Function. The permissive interlocks for reactor startup provide assurance that certain conditions desirable for startup are met. They, therefore, perform an administrative, rather than a safety, function.

d. Applicable Safety Analysis. No permissive interlocks are shown in Table VII of the HSR.[1] Although the permissive interlocks are listed as startup functions in Table III of the HSR addendum,[2] no analysis is presented in support of their inclusion in the RSS.

e. Safety Conclusions. It has been concluded that the permissive interlocks for reactor startup are not required in the RSS but should be retained to administratively assure required conditions for reactor startup. Accordingly, required plant conditions and minimum configurations have been defined for these functions and are designated Limiting Conditions for Operation in the EBR-II Technical Specifications.

f. Recommendations for System Upgrading. The Boland study (see Sect. III.B) recommends the relocation of the crane position, any control rod not down, and auxiliary pump operate interlock functions to non-RSS circuits, but no further action has been taken.

g. Summary of Modifications Performed or Proposed. Subsequent to publication of the HSR Addendum[2] and before May 1970, the crane position permissive interlock was added to the RSS.

As noted previously, fuel handling incomplete and any control rod unlatched were converted from reactor trip functions to permissive interlocks for reactor startup by PM 431 and PM 405, respectively.

Interlock circuitry for the three control rod positions converted to in-core test facilities was deactivated in conjunction with the plant modifications effecting the conversions.

h. Compliance with C16-1T. The retention of these functions as startup permissives in the RSS does not comply with Sect. 4.3.7, which disallows the use of protective functions to enforce administrative requirements. (It does, however, comply with Sect. 4.3.4, which specifies that signals from a system external to the PPS, i.e., control systems, may be supplied to PPS actuator circuits provided with proper isolation.)

16. Neutron Monitoring System (Unrestricted Fuel Handling Mode)

a. Description of Original System. The original system for neutron monitoring consisted of the following trips and sensors for unrestricted fuel handling:[2]

Source flux level (3 log count rate low-range nuclear channels designated 1, 2, and 3 in a 2/3 trip logic, may be bypassed for changeout of neutron source)

Log count rate level high (3 log count rate low-range nuclear channels designated 1, 2, and 3 in a 2/3 trip logic)

Period short (3 log count rate low-range nuclear channels designated 1, 2, 3; and 3 log N intermediate range nuclear channels designated 4, 5, and 6; each in a 2/3 trip logic)

Linear level (one linear intermediate range nuclear channel designated as 7, 1/1 trip logic)

High voltage low channel trip (one voltage detector per nuclear channel, low voltage causes trip on affected channel)

Linear level range setting--permissive interlock (amplifier on correct range, 1/1 circuit logic)

b. Description of Modified System. The modified system for neutron monitoring consists of the following trips and sensors for unrestricted fuel handling:

Count rate low (LCR subsystem of 3 wide-range nuclear channels designated A, B, and C in 1/3 trip logic, may be bypassed for changeout of neutron source)

Count rate high (LCR subsystem of 3 wide-range nuclear channels designated A, B, and C in 2/3 trip logic)

Period short (LCR subsystem of wide-range channels A, B, and C in 2/3 trip logic)

Linear level (two linear intermediate range nuclear channels designated 7 and 7A, 1/1 trip logic for either channel)

High voltage low channel trip (one voltage detector per nuclear channel, low voltage causes trip on affected channel)

Linear level range setting--permissive interlock (amplifier on correct range, 1/1 circuit logic for either channel)

c. Safety Concerns Related to Protective Function. Primary protection against fuel overtemperature as a result of reactivity insertion events during unrestricted fuel handling is provided by trips on period short and count rate high, with backup protection provided by trip on linear power level high. The specific reactivity events of concern involve administrative loading errors bringing the reactor to or near critical combined with loading or dropping a high-worth subassembly.

d. Applicable Safety Analysis. Appendix A, Sect. 1, and Appendix F, Sect. 1, of the HSR[1] and HSR Addendum,[2] respectively, report the original reactivity insertion safety analysis. Specifically, the following cases applicable to the fuel handling mode were considered: both safety rods, a central driver subassembly, or one control rod driven into a just-subcritical reactor. (Note that for the above to be pertinent as fuel-handling events, a prior error in loading to just-subcritical would have to be assumed.)

Although Table VII of the HSR[1] does not differentiate between reactor trip and safety rod trip, Section IV.A.2.c states: "Measurement of neutron flux is employed for three general purposes as follows:...Initiate automatic scram of control rods, or dropping of safety rods, whenever the power level exceeds a preset value or the reactor period becomes excessively short." Source flux level, log count rate level, high-voltage supply, period, and linear level and range are listed as fuel handling trips in Table III of the HSR Addendum.[2]

Section VIII-A of the EBR-II Status Report[3] provides an update of the original reactivity insertion analysis. In 1970, the HSR analysis of reactivity-related events was again reworked, using the then-existent PPS and improved computer techniques.[6]

The analytical document for reactivity-related protective functions for unrestricted fuel handling[22] identifies and addresses the following reactivity-related faults, classified as anticipated and unlikely in accordance with C16-1T[4] that were judged to define response time criteria and limiting setpoints for count rate, period, and linear power level trips:

Single subassembly loading error resulting in higher reactivity worth

Subassembly loading errors resulting in near-critical state (less than $1.8 subcritical)

Dropping of a single subassembly in the far subcritical state

Extremely unlikely faults involving the above conditions plus the loading or dropping of a high-worth subassembly were also investigated.

e. Safety Conclusions. None of the reactivity insertion events identified as either anticipated or unlikely result in fuel handling incidents requiring protective action; thus, to establish the protective functions required, the extremely unlikely event of loading a maximum-worth subassembly into a near-critical core at a high rate of speed was investigated and chosen as the limiting case for all reactivity-insertion events. For this event, trips on count rate level high and period short are required during unrestricted fuel handling with safety rods up. With safety rods down, an additional trip is recommended for a factor of 10 increase in count rate level. (With safety rods down, trip would initiate an interlock terminating gripper motion.)

Limiting conditions for operation during unrestricted fuel handling as defined in the EBR-II Technical Specifications apply to fuel handling with safety rods up and require count rate high and period short.

f. Recommendations for System Upgrading. None of the early studies specifically addresses fuel handling trips since they were limited in scope to conditions of reactor operation.

g. Summary of Modifications Performed or Proposed. Linear channel 7A was added to the system as backup for channel 7 by PM 230.

Channels 1, 2, 3, 4, 5, and 6 were replaced by wide-range nuclear channels A, B, and C by PM WAF-753.

h. Compliance with C16-1T. With the exception of linear level channels 7 and 7A, the modified neutron monitoring system meets the requirements of C16-1T with the variance noted in Sect. IV.C.4.h. Fuel handling trip circuitry has not been upgraded to comply with C16-1T.

17. Safety Rods Up Monitor (Fuel Handling Mode)

a. Description of Original System. The original system for safety rods up monitoring consisted of the following interlock trips and sensors for unrestricted fuel handling:[2]

> Safety rods not in "full up" position (2 limit switches, 1/2 trip logic, may be bypassed during unrestricted fuel handling for safety rod changeout)

b. Description of Modified System. The basic function of the system for safety rods up monitoring has not been modified.

c. Safety Concerns Related to Protective Function. The postulated events of concern relative to safety rods up are those same TOP events identified in Sect. 16. Fuel handling with safety rods up (in core) assures negative reactivity to offset reactivity insertion incidents. The safety rods up interlock thus performs an administrative function to assure that safety rods are operable when needed.

d. Applicable Safety Analysis. The TOP safety analysis pertinent to unrestricted fuel handling with safety rods up may be found in Ref. 1, 2, 3, 6, and 22 (see Sect. IV.c.16).

Although trip on safety rods not up is not listed in Table VII of the HSR,[1] there are several references in the text to the fact that unrestricted fuel handling may not be performed with the safety rods down. For example, in Sect. III.A.6., it is stated that

"...The safety rods are always in the reactor...The primary purpose of the safety rods is to provide 'available negative reactivity' when the reactor is shut down and the control rods are disconnected. They provide a safety factor during reactor loading operations." Section III.A.6.b states "...All reactor operations, including...actuation of the fuel handling system, require the safety rods to be in the up position."

Although trip on safety rods not fully up is listed in Table III of the HSR Addendum,[2] no analysis is presented in support of its inclusion in the RSS.

e. Safety Conclusions. It has been concluded that the trip on safety rods not fully up is not required in the RSS and that unrestricted fuel handling may be safely performed with the safety rods down (out of core), provided that gripper motion is terminated upon period and level trip. The limiting case is the unlikely fault of a combination of loading errors that increase reactivity nearer to critical than the replacement worth of a single central subassembly, thus increasing vulnerability to a criticality event of the extremely unlikely category involving concurrent dropping or lowering of a high-worth subassembly in the core.

The analysis of Ref. 22 has only recently been concluded. Consequently, the EBR-II Technical Specifications currently require that unrestricted fuel handling be conducted with safety rods up. Should a proposal be prepared to delete the trip on safety rods not up from the

RSS, the appropriate limiting conditions for operation will be developed for unrestricted fuel handling with safety rods down.

f. Recommendations for System Upgrading. The safety evaluation for unrestricted fuel handling recently completed supports deletion of the trip on safety rods not fully up from the RSS, but no such recommendation has been made.

g. Summary of Modifications Performed or Proposed. Removal of the trip on safety rods not up from the RSS (unrestricted fuel handling mode) and modification of the trip function to terminate gripper motion only was proposed as part of PM 443, a modification to accommodate fuel handling with safety rods down. PM 443 was subsequently cancelled, and no proposal to delete this trip from the RSS is planned at this time.

To avoid excessive spurious trips during the initial and final stages of unrestricted fuel handling when reactivity-insertion events are not possible, a safety rods up bypass was installed by PM 226. A similar bypass for use during rotating plug seal cleaning was installed by PM 338.

h. Compliance with C16-1T. The retention of this trip does not comply with C16-1T which disallows the retention of unnecessary trips.

18. Interlock Circuits for Unrestricted Fuel Handling

a. Description of Original System. The original system contained the following interlock circuits for unrestricted fuel handling:[2]

> Source flux level (3 log count rate low-range nuclear channels designated 1, 2, and 3 in a 2/3 trip logic, may be bypassed for changeout of neutron source)--see Sect. 16*
>
> Linear level range setting (amplifier on correct range, 1/1 circuit logic)--see Sect. 16*
>
> Any control rod not in "full down" position (limit switch for each rod, 1/9 circuit logic, bypassed when reactor vessel cover is raised)
>
> Safety rods not in "full up" position and "hand operated" drive interlock (limit switch for each rod, may be bypassed for changeout of safety rod)--see Sect. 9*
>
> Power to MG on: primary pumps 1 and 2 (power off to either pump 1/2 circuit logic)

b. Description of Modified System. The modified system contains the same interlock circuits for unrestricted fuel handling.

c. Safety Concerns Related to Protective Function. The any-control-rod-not-down permissive interlock for unrestricted fuel handling provides assurance that a condition desirable for fuel handling is met. It thus performs an administrative, rather than a safety, function. The postulated event of concern relative to pump power on is the core disruption that could occur if primary flow were begun.

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*Those functions addressed in other sections will not be discussed further in this section.

d. Applicable Safety Analysis. None of the fuel handling interlocks is mentioned in the HSR.[1] Although these interlocks are shown in Table III of the HSR Addendum,[2] no supporting analysis is provided in support of their inclusion in the RSS.

e. Safety Conclusions. It has been concluded that none of the fuel handling interlocks are required in the RSS. Hydraulic hold-downs preclude a core disruption event on initiation of primary flow during fuel handling. However, it was further concluded that these interlocks should be retained to administratively assure required conditions for unrestricted fuel handling. Accordingly, required plant conditions and minimum configurations have been defined for these functions and are designated Limiting Conditions for Operation in the EBR-II Technical Specifications.

f. Recommendations for System Upgrading. None of the early studies specifically addresses fuel handling interlock functions, since spurious fuel handling trips were not the object of concern.

g. Summary of Modifications Performed or Proposed. No modifications to this system have been performed or are proposed.

h. Compliance with C16-1T. The retention of these interlock functions for unrestricted fuel handling in the RSS does not comply with Sect. 4.3.7, which disallows the use of protective functions to enforce administrative requirements. (However, it does comply with Sect. 4.3.4, which specifies that signals from a system external to the PPS, i.e., control systems, may be supplied to PPS actuator circuits provided with proper isolation.)

19. Manual Trip Function

a. Description of Original System. The original system contained the following manual trip functions for reactor operation and/or unrestricted fuel handling:[2]

Control rod manual (one pushbutton on reactor console, trips safety rods as well, 1/1 trip logic)

Safety rod manual (two pushbuttons, one on reactor console and one on fuel handling console, 1/2 trip logic)

b. Description of Modified System. The modified system contains the same manual trip functions for reactor operation (Systems A and B) and/or unrestricted fuel handling.

c. Safety Concerns Related to Protective Function. The postulated events of concern relative to manual trip functions are those postulated events of sufficient severity to warrant operator action but for which automatic trip is not justified nor required. Further, the manual trip function provides backup in the event of automatic trip function failure.

d. Applicable Safety Analysis. Manual trip functions are not mentioned in the HSR.[1] Although the manual trip functions are shown in Fig. 35 of the HSR Addendum,[2] no supporting analysis is provided for their inclusion in the RSS.

The analysis of events leading to bulk sodium temperature high[18] and delayed neutron signals[21] addresses the need for manual trip in certain cases.

e. Safety Conclusions. It was concluded that the manual trip functions are required for reactor operation and unrestricted fuel handling. Accordingly, minimum configurations have been defined for these functions and are designated Limiting Conditions for Operation in the EBR-II Technical Specifications.

f. Recommendations for System Upgrading. No recommendations for system upgrading have been made.

g. Summary of Modifications Performed or Proposed. No modifications to this system have been performed or are proposed.

h. Compliance with RDT C16-1T. The system provides the manual means to meet the requirements of Sect. 4.2.6.

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ARGONNE
NATIONAL
LABORATORY

INTRA-LABORATORY MEMO

May 25, 1977

To: R. G. Sachs Laboratory Director

Attn: C. A. DeLorenzo Director, OOS

From: D. W. Cissel Director, EBR-II Project

Subject: Summary Report on EBR-II PPS Upgrade

Reference: (1) Letter, R. H. Bauer to R. G. Sachs, same subject, May 12, 1976

(2) Letter, G. H. Golden to M. E. Jackson, subject: "Information Milestones for EBR-II Analysis Department," November 1, 1976

Attached for transmittal to ERDA-CH are copies of the report entitled, "A Summary of Modifications to the Original EBR-II Reactor Shutdown System," by N. L. Gale. This report was prepared in response to Ref. 1 and satisfies a Project milestone commitment for May, 1977 (Ref. 2).

Sufficient copies have been included for distribution to the RSRC for information only. Since the document is basically a summary of material previously reviewed and approved by both the RSRC and ERDA-CH, RSRC review is not required.

DWC/NLG:n

Enclosures 30

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